Exelon Generation.

June 1, 2015

10 CFR 50.73

SVP-15-043

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Quad Cities Nuclear Power Station, Unit 1
Renewed Facility Operating License No. DPR-29
NRC Docket No. 50-254

Subject: Licensee Event Report 254/2015-005-00, "Manual Scram Due to Steam Leak on Turbine Throttle Pressure Sensing Line"

Enclosed is Licensee Event Report (LER) 254/2015-005-00, "Manual Scram Due to Steam Leak on Turbine Throttle Pressure Sensing Line," for Quad Cities Nuclear Power Station, Unit 1.

This report is submitted in accordance with 10 CFR 50.73 (a)(2)(iv)(A) which requires the reporting of any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B).

There are no regulatory commitments contained in this letter.

Should you have any questions concerning this report, please contact Mr. W. J. Beck at (309) 227-2800.

Respectfully,

Scott Darin
Site Vice President
Quad Cities Nuclear Power Station

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station

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NRR

NRC FORM 366 (02-2014) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Quad Cities Nuclear Power Station Unit 1	05000254	1 OF 5

4. TITLE

Manual Scram Due to Steam Leak on Turbine Throttle Pressure Sensing Line

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME: N/A	DOCKET NUMBER: N/A
04	02	2015	2015 -	005 -	00	06	01	2015	FACILITY NAME: N/A	DOCKET NUMBER: N/A

9. OPERATING MODE: 1

10. POWER LEVEL: 20

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Tom Petersen – Regulatory Assurance	(309) 227-2825

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	SB	N/A	N/A	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO		N/A	N/A	N/A

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On April 2, 2015, at 1810 hours, with Unit 1 in Mode 1 and operating at 100% power, an alarm was received in the control room indicating that two of three turbine throttle pressure transmitters on Unit 1 had failed low. Reactor power was reduced to 20% and the main turbine was secured. A steam leak was confirmed to have originated between the turbine throttle pressure transmitter sensing line isolation valve and the 30 inch main steam D-Ring header. Since the steam leak could not be isolated with Unit 1 at power, a manual scram was inserted on Unit 1 at 2133 hours, and the main steam isolation valves (MSIVs) were manually closed. A forced outage was initiated to investigate and perform repairs.

The reactor and turbine responded as designed. Operators performed required actions safely and in accordance with procedures and training. Operators needed to use relief valves to control reactor pressure since the turbine bypass valves were unavailable for normal pressure control after closing the MSIVs to isolate the steam leak at the D-ring. A drywell pressure rise occurred due to a relief valve tailpipe vacuum breaker which had inadvertently stuck open; the relief valve was then closed, a drywell cooler was started, and drywell pressure returned to normal.

The root cause of the failure of the sensing line was determined to be inadequate monitoring of the sensing line supports that allowed each of the clamps to degrade and loosen over time resulting in high cycle fatigue cracking and eventual fracture of the sensing line.

Corrective actions included replacing the 3/4 inch carbon steel section of the failed sensing line, including the isolation valve, and replacing the associated degraded supports. Future corrective actions include conducting focused inspections of small bore piping/tubing on both accessible and inaccessible systems.

The safety significance of this event was minimal. This report is submitted in accordance with 10 CFR 50.73 (a)(2)(iv)(A) which requires the reporting of any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B).

NARRATIVE

PLANT AND SYSTEM IDENTIFICATION
General Electric - Boiling Water Reactor, 2957 Megawatts Thermal Rated Core Power

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

EVENT IDENTIFICATION

Unit 1 Manual Scram Due to Un-Isolatable Steam Leak on Turbine Throttle Pressure Sensing Line

A. CONDITION PRIOR TO EVENT

Unit: 1	Event Date: April 2, 2015	Event Time: 2133 hours
Reactor Mode: 1	Mode Name: Power Operation	Power Level: 20%

B. DESCRIPTION OF EVENT

On April 2, 2015, at 1810 hours, with Unit 1 in Mode 1 and operating at 100% power, an alarm [PA] was received in the control room [NA] indicating that two of three turbine [TA] throttle pressure transmitters [PT] on Unit 1 had failed low. Shortly after receiving the alarm, the control room was notified of an apparent steam leak in the Unit 1 low pressure heater bay (LPHB) based on loud noises emanating from several heater bay access door [DR] areas. Operations was dispatched to the LPHB to investigate the potential steam leak and possible reason for the throttle pressure alarms. A steam leak originating in the area of the main steam [SB] equalizing header (D-Ring) was confirmed although the exact location of the leak could not be immediately determined due to the conditions. As a result, Operations began an emergency power reduction on Unit 1 to 20% reactor power and the main turbine was secured in an initial attempt to isolate the steam from the unidentified leak location. With the main turbine tripped another LPHB entry was performed which confirmed that the steam leak originated between the turbine throttle pressure transmitter sensing line isolation valve [ISV] (1-3099-110B) and the 30 inch main steam D-Ring header. With the knowledge that the steam leak could not be isolated with Unit 1 at power, Operations inserted a manual scram from 20% reactor power on Unit 1 at 2133 hours on April 2, 2015, and manually closed the main steam isolation valves (MSIVs).

On April 3, 2015, with the steam effectively isolated to the D-Ring, an inspection of the fractured sensing line (1-30215B-3/4) and related components was performed. In addition to the sensing line failure, it was identified that each of the supports [SPT] associated with the failed sensing line (three total) were either no longer attached or were found loose. One of the three supports, a Unistrut cross brace located closest to the fracture location, was found on the floor with the pipe clamps still attached to the pipe and severely damaged. A second clamp, further away from the fracture location, was found detached from its Unistrut and also lying on the floor and a third clamp, furthest away from the fracture location, was found with a loose nut on the bolting that provides the clamping load.

The failed sensing line and the isolation valve (1-3099-110B) were replaced and the new socket welds were upgraded to the Electric Power Research Institute (EPRI) recommended 2:1 configuration to enhance the small bore piping assembly's resistance to high cycle fatigue. The three degraded supports associated with the sensing line were also replaced. The adjacent "A" sensing line feeding the turbine throttle pressure transmitter (1-30215A-3/4) which was constructed similarly to the failed "B" sensing line had a liquid penetrant test (PT) performed on its socket welds and no fatigue cracking was evident.

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Additionally, other small bore piping (less than or equal to 2 inches) connected to the main steam header (25 total) were inspected for evidence of similar vibration related degradation. All issues identified during the walkdown (including additional loose supports) were corrected prior to startup.

On April 3, 2015, at 0032 hours, ENS #50949 was made to the NRC under 10 CFR 50.72 (b)(2)(iv)(B), RPS actuation while critical, and 10 CFR 50.72 (b)(3)(iv)(A), and valid specific system actuations (MSIV closures).

The repairs were completed on April 4, 2015, and the re-start of Unit 1 commenced at 0625 hours on April 5, 2015.

The safety significance of this event was minimal. This event is reportable per 10 CFR 50.73(a)(2)(iv)(A), which requires the reporting of any event or condition that resulted in manual or automatic actuation of the reactor protection system (RPS) including reactor scram, MSIV closures, and 1C drywell fan cooler operation.

C. CAUSE OF EVENT

The root cause of the failure of the sensing line was determined to be inadequate monitoring of the sensing line supports that allowed each of the clamps to degrade and loosen over time resulting in high cycle fatigue cracking and eventual fracture of the sensing line.

The failed sensing line consisted of a short 3/4 inch section of carbon steel pipe welded to the top of a 30 inch diameter main steam equalizing header (D-ring). An isolation valve (1-3099-110B) was welded to the 3/4 inch pipe and the opposite end of the isolation valve was welded to a long run of 1/2 inch diameter stainless steel piping that was connected to the pressure transmitter for that sensing line. The sensing line traveled vertically upward from the D-ring header for approximately 6 feet and then traveled horizontally for more than 20 feet before traveling vertically downward to the next elevation where it exited the LPHB and connected into the pressure transmitter. This long run of sensing line piping had been adequately supported during the previous 42 years of plant operation as demonstrated by the successful performance of the sensing line even during periods of higher main steam line (MSL) vibrations in the history of the plant. The total of three (3) supports effectively increased the rigidity of the sensing line, thereby limiting the effects of the MSL vibrations.

Over time, the supports associated with the failed sensing line began to degrade due to normal piping vibration. All three supports (one on the vertical section and two on the horizontal section of the piping run) had loosened over time and were no longer performing their intended function at the time of the event. As the plant started up from refueling outage Q1R23, the normal MSL vibrations caused the now inadequately supported sensing line to experience high cycle fatigue. The fatigue crack originated on the outside diameter of the 3/4 inch carbon steel pipe in an area adjacent to the socket weld connecting the 1-3099-110B isolation valve to the pipe.

D. SAFETY ANALYSIS

System Design

The main steam piping consists of four 24-inch diameter lines extending from the outermost MSIVs to the main stop valves of the turbine. All four MSLs are connected by a 30-inch diameter main steam equalizing header. The sensing line that failed connects to the main steam equalizing header for the purpose of supplying header sensing pressure to two of three digital electro-hydraulic control [JJ] (DEHC) system throttle pressure transmitters.

The DEHC system pressure regulator and turbine-generator controllers utilize a triple modular redundant (TMR) design. The controllers are integrally connected to accomplish the functions of controlling reactor pressure and turbine speed. The controllers utilize triple-redundant process sensors and will continue operation if one of the

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process sensors fail, which includes the throttle pressure transmitters for the pressure controller. The pressure controller is designed to continue operating even if two of the three sensors fail.

Safety Impact

The safety significance of this event was minimal. The reactor [AC] scram and turbine trip responded as designed. All safety functions performed as expected including the control rod drives [AA] and primary containment isolation [JM] functions. Operators performed required actions safely and in accordance with procedures and training. Operators needed to use relief valves [RV] to control reactor pressure since the turbine bypass valves were unavailable for normal pressure control after closing the MSIVs to isolate the steam leak at the D-ring. However, during the reactor cooldown when the 3B relief valve was opened, a corresponding unexpected drywell [NH] pressure rise occurred. The 3B relief valve was then closed and the 1C drywell cooler [VB] was started to lower the drywell pressure rise; parameters then returned to normal. The drywell pressure rise was caused by the 3B relief valve tailpipe vacuum breaker [BF] which had inadvertently stuck open.

Risk Insights

A risk assessment of the manual scram due to the steam leak and subsequent unexpected drywell pressure rise was performed utilizing the Plant Probabilistic Risk Assessment (PRA). In conclusion, the steam leak originating between the Turbine Throttle Pressure Transmitter sensing line isolation valve (1-3099-110B) and the 30 inch Main Steam D-Ring header, when coupled with the inadvertently stuck open 3B ERV tailpipe vacuum breaker, was determined to be not risk significant.

E. CORRECTIVE ACTIONS

Immediate:

1. The 3/4 inch carbon steel section of the sensing line, including the isolation valve, was replaced. The new socket welds were upgraded to the EPRI recommended 2:1 configuration.
2. The degraded supports associated with the failed sensing line were replaced. The support located on the vertical piping run was relocated to the horizontal run to mitigate potential long-term slippage due to gravity.
3. The 3B ERV tailpipe vacuum breaker was repaired.

Follow-up:

1. Focused inspections of small bore (less than or equal to 2 inch) piping/tubing will be conducted on both accessible and inaccessible systems.
2. ERV vacuum breaker preventive maintenance procedure to be revised to include a disk hinge bearings test.

F. PREVIOUS OCCURRENCES

The station events database, LERs, and INPO Consolidated Event System (ICES) were reviewed for similar events at the Quad Cities Nuclear Power Station. This event was a manual reactor scram due to a steam leak attributed to inadequate monitoring of the sensing line piping supports that allowed each of the clamps to degrade and loosen over time resulting in high-cycle fatigue cracking and eventual fracture of the sensing line. No similar events were identified where piping supports became degraded resulting in high cycle fatigue. Based on the cause of this event and the associated corrective actions, the events listed below, although similar in topic, are not considered significant station experiences that would have directly contributed to preventing this event.

- LER 254/2011-002-00, 08/12/11, Unit 1 Manual Reactor Scram Due to Steam Leak (06/13/11) - A newly installed sensing line located on a 20 inch main steam header failed due to high cycle fatigue, resulting in a steam leak and a manual scram. The root cause was that the capped pipe stub repair failed due to a fatigue induced flaw that was not known to exist at the time of the repair. This previous event was caused by inadequate design rigor associated with the new installation and pre-existing flaws within the weld contributed to the failure, whereas this

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current LER event was caused by piping supports that became degraded resulting in high cycle fatigue cracking and eventual fracture of the sensing line.

G. COMPONENT FAILURE DATA

Failed Equipment: Sensing Line (1-30215B-3/4)
Component Manufacturer: Commercially available
Component Model Number: Non-safety related, 3/4 inch, schedule 80, ASTM A-106 GR. B carbon steel piping
Component Part Number: N/A

Failed Equipment: Degraded supports associated with the sensing line
Component Manufacturer: Unistrut
Component Model Number: Non-safety related, 1-5/8 inch by 1-5/8 inch, 12 gauge, ASTM A-446 GR. A carbon steel channels and associated clamps
Component Part Number: N/A

This event has been reported to ICES as Failure Report No. 316001.